Ignited Spherical Tokamaks and their place in fusion[1]

L. E. Zakharov[1],

[1] *Princeton Plasma Physics Laboratory, MS-27 P.O. Box 451, Princeton NJ 08543-0451,*

presented by Leonid E. Zakharov,

Culham Physics Seminar,

January 20, 2004, Culham, UK

[1] This work is supported by US DoE contract No. DE–AC020–76–CHO–3073.

Abstract

Three basic aspects of the reactor physics and technology, i.e., Operational Power Reactor Regime (OPRR), development of the First Wall (FW), and self-sustained Tritium Cycle (TC) are discussed in the talk.

The notion of OPRR is introduced explicitly in order to distinguish it from the relatively short ignition phase of the reactor operation. In contrast to ignition, OPRR requires new confinement and stability regimes with high beta (>8 %) and relatively small confinement time (<1.5 sec). Being a challenge for the plasma physics, OPRR cannot be developed without use of the fusion power. At the same time, the physics and technology of FW and TC cannot be developed without OPRR.

This generic link between 3 key elements of the reactor physics together with consumption of large amount of tritium for their development, creates a gap on the development path toward the magnetic fusion reactor.

In this regard, Ignited Spherical Tokamaks (IST) seem to be the only feasible concept for bridging this gap between the present physics and future power reactors.

Contents

There are 3 key objectives specific to magnetic fusion:

1. Achieving Ignition and Operational Power Reactor Regime (OPRR),
2. Design and development of the low activation First Wall (FW) together with power extraction and helium ash exhaust,
3. Tritium Cycle (TC),

all compatible with safety and economics of the power reactors.

Besides a short phase of Ignition

all other 3 components are generically linked to each other

A fusion reactor should be able to reach the “ignition”

$$\frac{E_{pl}}{\bar{\tau}_E} = f_\alpha \int_{V_0} P_\alpha dV, \quad P_\alpha = E_\alpha n_D n_T \langle \sigma v \rangle_{DT}, \qquad (1.1)$$
$$\frac{E_{pl}}{\bar{\tau}_E} \equiv \frac{E_{pl}}{\tau_E} + \int_{V_0} P_{rad} dV,$$

where

- E_{pl} - *total plasma energy,*
- V_0 - *total plasma volume,*
- P_α - *density of the* α*-particle power deposition,*
- P_{rad} - *radiation power density,*
- E_α - *3.5 MeV,*
- n_D, n_T - *densities of deuterium and tritium,*
- $\langle \sigma v \rangle_{DT}$ - *cross-section of the reaction,*
- f_α - *fraction of used* α*'s,*
- τ_E - *energy confinement in plasma physics sense,*
- $\bar{\tau}_E$ - *overall energy confinement (accounting for radiation).*

An appropriate form of P_α

$$\frac{1}{V_0}\int P_\alpha dV = C_\alpha \left\langle 4p_D p_T \right\rangle = \left\langle p \right\rangle^2 f_{pk} C_\alpha,$$
$$C_\alpha \equiv \frac{\langle P_\alpha \rangle}{\langle 4p_D p_T \rangle}, \quad f_{pk} \equiv \frac{\langle 4p_D p_T \rangle}{\langle p \rangle^2}, \tag{1.2}$$

where

$\langle \dots \rangle$ - *stands for volume averaging,*

E_{pl} - *total plasma energy,*

p, p_D, p_T - *plasma,deuterium and tritium ions pressure,*

C_α - *reactivity factor, depending on* T, p_D, p_T *profiles,*

f_{pk} - *"peaking" factor, taking into account peakedness of the plasma pressure profile, dilution of DT mix by helium ash and by impurities, and the difference in electron and ion temperatures.*

Reactivity factor C_α depends on the plasma profiles and $< T >$

Fig. 1 shows C_α for different peakedness of plasma temperature and density

$$
\begin{aligned}
n_e(V) &= \langle n_e \rangle \, s_{\nu_n}(\bar{V}), \quad T(V) = \langle T \rangle \, s_{\nu_T}(\bar{V}), \\
s_\nu(\bar{V}) &\equiv (1+\nu)(1-\bar{V})^\nu, \quad \bar{V} \equiv \frac{V}{V_0}.
\end{aligned} \tag{1.3}
$$

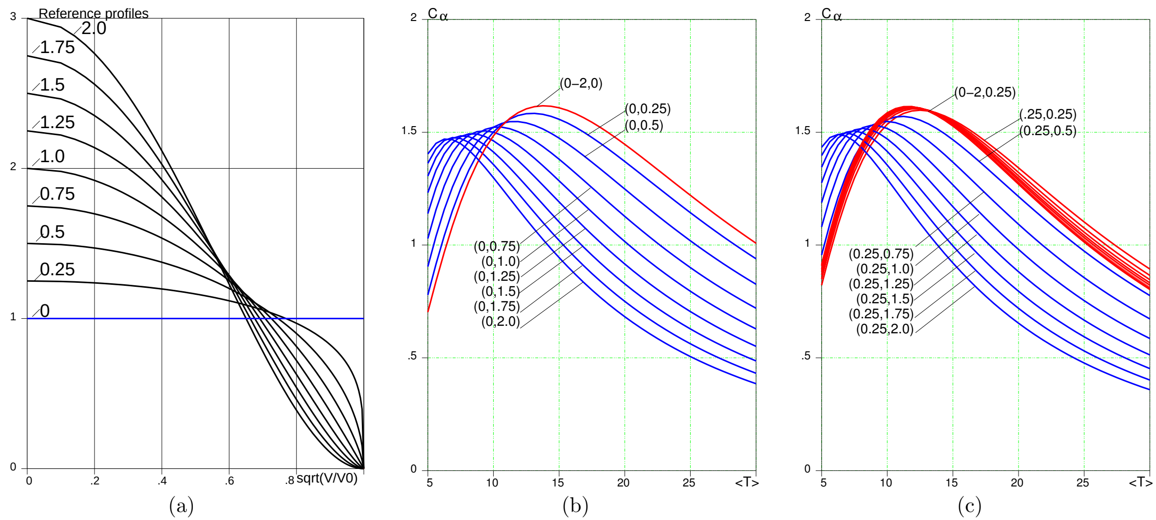
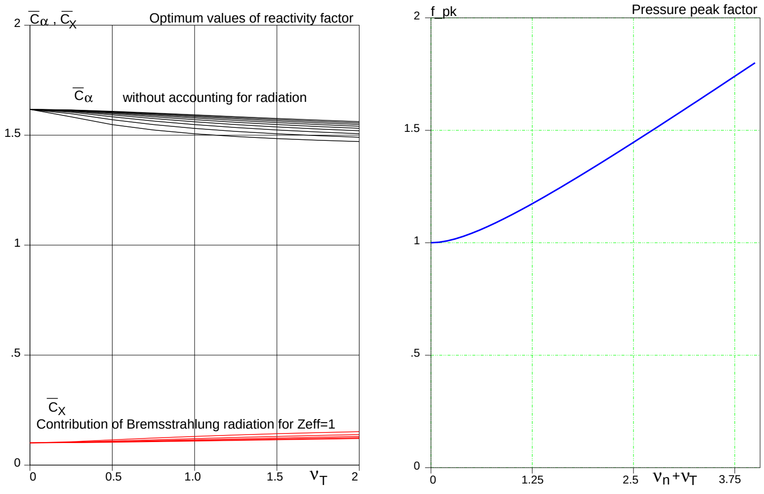


Figure 1: (a) Reference profiles (with the same volume averaged values) for plasma density and temperature. (b) C_α as a function of averaged plasma temperature. The red curve for $\nu_T = 0$ does not depend on $\nu_n = 0-2$. Blue curves correspond to $\nu_n = 0$ and $\nu_T = 0.25-2$. (c) C_α for another set of profiles. Red curves are for $\nu_T = 0.25$ with $\nu_n = 0.25-2$, and blue curves are for $\nu_n = 0.25$ with $\nu_T = 0.25-2$.

Optimal reactivity factor $\bar{C}_\alpha \simeq 1.5$ is insensitive to plasma profiles



(a) (b)

Fig. 2 (a) Optimum reactivity factor $\bar{C}_\alpha$ of alpha particles and Bremsstrahlung radiation factor $\bar{C}_X$ ($Z_{eff}=1$) for different reference density and temperature profiles (ν_n=0-2,ν_T=0-2) at optimum plasma temperature. (b) Pressure peaking factor for reference profiles.

Ignition criterion in terms of C_α

$$\frac{3}{2}\frac{\langle p\rangle}{\bar{\tau}_E} = f_\alpha \langle P_\alpha\rangle ,$$

$$\frac{2}{3} C_\alpha f_\alpha f_{pk} \langle p\rangle \bar{\tau}_E =, 1$$

or in terms of optimal $\bar{C}_\alpha$

$$f_{pk}\langle p\rangle f_\alpha \bar{\tau}_E = 1,$$

or

$$f_{pk}\langle p\rangle \bar{\tau}^*_E = 1, \qquad \bar{\tau}^*_E \equiv f_\alpha \bar{\tau}_E$$

Equivalent forms

$$f_{pk}\langle n_e T\rangle \bar{\tau}^*_E = 31\cdot 10^{20}, \quad f_{pk}\beta B^2 \bar{\tau}^*_E = 2.5, \quad \beta \equiv \frac{2\mu_0\langle p\rangle}{B^2}, \quad n_0 T_0 \tau_E = 50\cdot 10^{20}. \tag{1.4}$$

$f_{pk}\langle p\rangle \bar{\tau}^*_E = 1$ should be fulfilled during both ignition phase and power production operation.

Fusion power density in a reactor is proportional to $\langle p\rangle^2$ and to τ_E^{-2}

Equation

$$P_{DT} = 5\int P_\alpha dV = 7.5 V_0 f_{pk}\langle p\rangle^2 = 1.2 V_0 f_{pk}(\beta B^2)^2 \tag{1.5}$$

determines the operational power reactor regime, e.g.:

$$\begin{aligned} &\beta = 0.1, \quad B = 5, \quad p = 1, \quad f_{pk} \simeq \frac{4}{3}, \quad P_{DT} = 10V_0, \\ &V_0 = 400 \rightarrow P_{DT} = 4000, \quad \bar{\tau}_E^* = 0.75 \end{aligned} \tag{1.6}$$

With the plasma pressure in the range of $0.8-1$ MPa the (overall) energy confinement time

$$\bar{\tau}_E = \frac{1}{f_{pk}\langle p\rangle}, \quad P_{DT} = 7.5\frac{V_0}{f_{pk}\bar{\tau}_E^{*2}} \tag{1.7}$$

is in the range of 0.8-1.3 sec.

Not the large $\bar{\tau}_E^*$, but its power dependence $\bar{\tau}_E^* \propto 1/\sqrt{P_\alpha}$ is essential for OPRR.

At $\bar{\tau}_E^* \simeq \tau_\alpha$ (slowdown time) some enhancement in conventional τ_E is necessary.

Ignition parameters are determined by the auxiliary heating power P_{ext}.

Transition into ignited phase

$$\frac{dE_{pl}}{dt} = P_{ext} - \frac{E_{pl}}{\bar{\tau}_E} + f_\alpha P_\alpha > 0. \tag{1.8}$$

In the best possible scenario: (a) optimal temperature profi le, (b) P_α is controlled by the density level,

$$\begin{aligned} \bar{\tau}_{E@ign}\frac{d\bar{E}}{dt} &= \frac{P_{ext}}{f_\alpha P_{\alpha@ign}} - \bar{E}\frac{\bar{\tau}_{E@ign}}{\bar{\tau}_E} + \bar{E}^2 > 0, \\ \bar{E} &\equiv \frac{E_{pl}}{E_{pl@ign}}, \quad P_\alpha = \frac{E_{pl}^2}{E_{pl@ign}^2} P_{\alpha@ign} = \bar{E}^2 P_{\alpha@ign}. \end{aligned} \tag{1.9}$$

The necessary power for ignition

$$P_{ext} > \frac{1}{4} f_\alpha P_{\alpha@ign} = \frac{1}{20} f_\alpha P_{DT@ign}, \quad \bar{\tau}_{E@ign} > \sqrt{\frac{1.5V_0}{4f_\alpha P_{ext}}}. \tag{1.10}$$

Ignition at operational point is impractical. E.g., $P_{DT} = 4$ GW would require $P_{ext} > 200$ MW.

High $\bar{\tau}_E$ (and low β) is required of ignition.

Ignition and OPRR have well separated $\bar{\tau}_E^*$'s in the power reactor

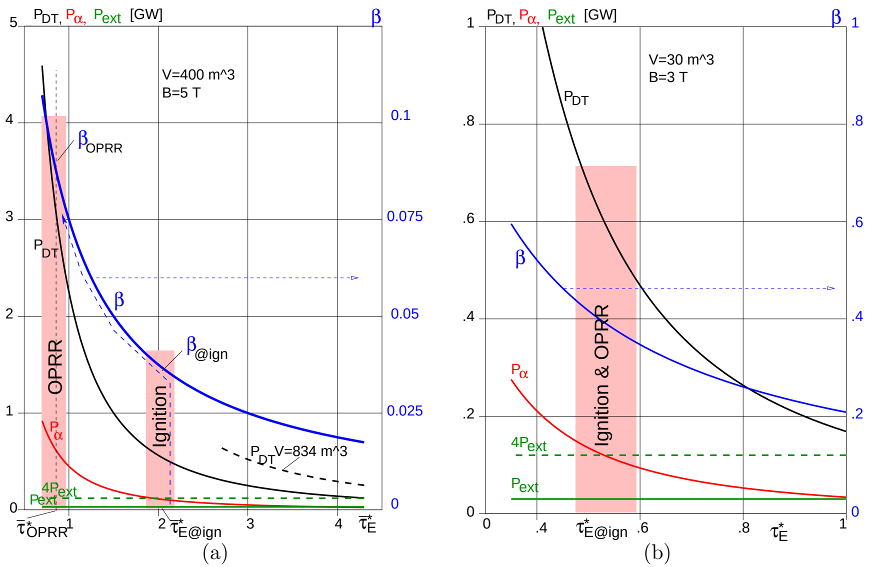


Figure 2: Fusion power vs cumulative energy confinement time $\bar{\tau}_E^*$ for (a) a reactor with $B = 5$ T and $V = 400$ m^3, and (b) for an Ignited Spherical Tokamak with $B = 3$ T and $V = 30$ m^3.

The high pressure plasma of OPRR ($\langle p \rangle \simeq 0.8 - 1$ MPa) can be developed only with use of fusion power as the dominant heat source.

OPRR and ignition parameters can coincide only in ST, with a small volume $V_0 \simeq 30$

Any cost estimates for fusion lead to a challenge

The monetary value of the electricity produced W_{Electr} during the life time of the reactor is limited. Assuming 30 years of uninterrupted operation, a reference estimate can be written as

$$W_{Electr}\ [\$B] = 10.5 P_{Electr} \cdot \frac{C_{kWh}}{0.04}, \tag{1.11}$$

where P_{Electr} [GW] is the electric power of the reactor, e.g., $P_{Electr} \simeq P_{DT}/4$, and C_{kWh} is the cost of 1 kWh.

The cost of the reactor should be a fraction of W_{Electr}.

Even such a extremely simplified estimate imposes severe restrictions.

A calibration with $5 B cost of the ITER (0.4 GW) shows that

the conventional plasma does not fi t even the simplest cost considerations.

FW$\equiv$(first 10-15 cm) is the most challenging component of the reactor

The characteristic neutron fluence for the FW life time is $\simeq$15 MW·year/m^2. It can be converted into the corresponding value C_{FW} of electricity "produced" per 1 m^2 during the life time of the FW element

$$C_{FW} \quad \left[\frac{\text{\$B}}{\text{m}^2}\right] \simeq 0.001 \cdot \frac{5.25}{4} \cdot \frac{C_{kWh}}{0.04}, \tag{1.12}$$

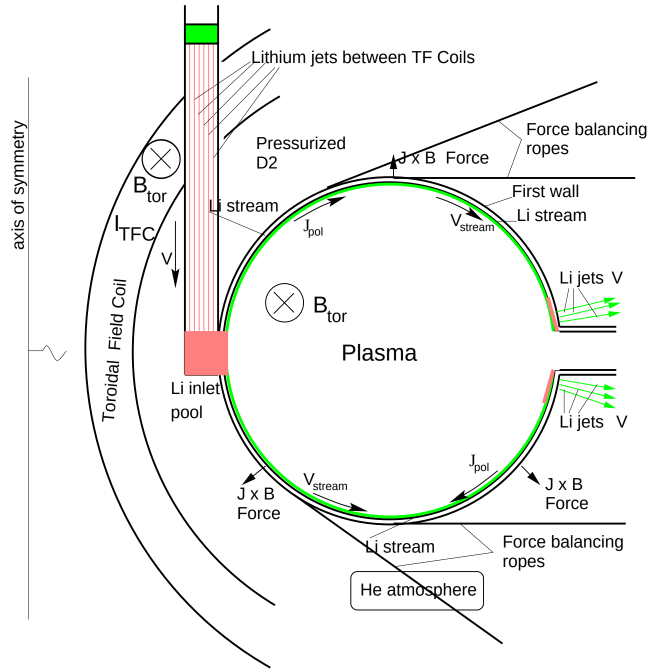
where $1/4$ is an assumed conversion factor of fusion power to electricity.

The cost of replacement of the first wall surface should be within the limit C_{FW} given by Eq. (1.12).

New approaches are strongly motivated for the FW design with emphasis on low activation structures and liquid elements (liquid lithium, FLiBe, Be, etc).

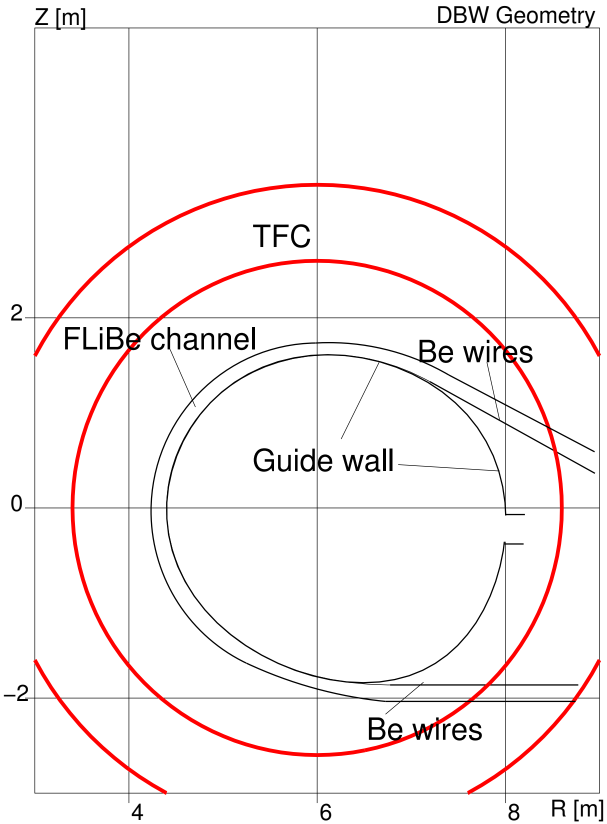
Intense Li Streams affect the very fundamentals of reactor design.

Electrodynamic pressure creates a stable situation for the first wall.



- Guide wall works against expansion $\Longrightarrow$
- Guide wall can be made as a thin shell (like a car tire).
- Inner surface is sealed by the lithium streams (insensitive to cracks) $\Longrightarrow$
- Vacuum barrier can be moved to the plasma boundary (giving access to the neutron zone).

Topology of Be wires can be made consistent with the presence of the FLiBe Blanket



Equation for poloidal curvature of the guide wall

$$\boxed{d\frac{T}{\rho} = p_{JxB} - p_{ext} - g\rho_{FLiBe}(z - z_0).}$$

Both radial force on both lines of wires

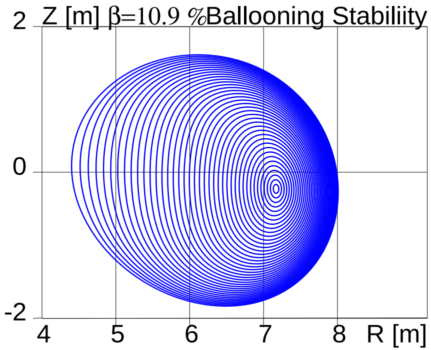
$$F = 1.5 \text{ [MN/m]}$$

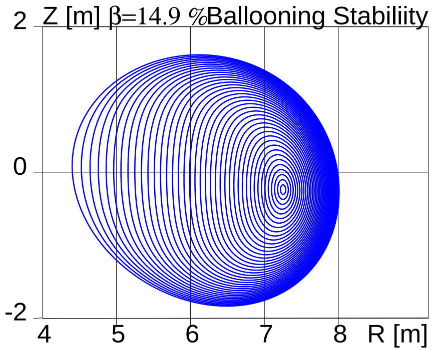
and tension in wires

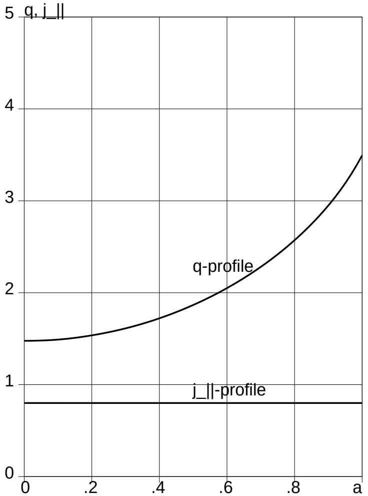
$$d \cdot T = 0.75 \text{ [MPa} \cdot \text{m]}$$

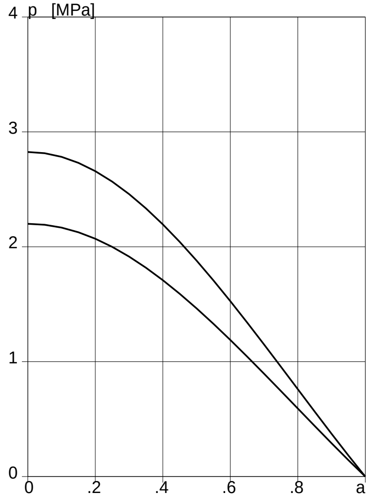
are reasonable.

Plasma shape is consistent with the wall stabilized high-β.

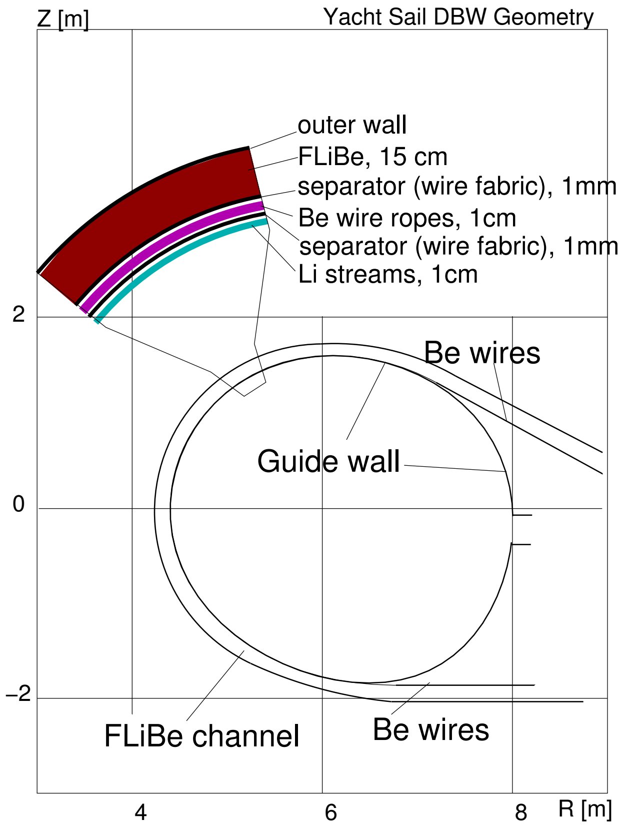








Yacht Sail FW design concept is the only one consistent with the FPR



- Intense Li Streams keep low temperature of the FW plasma side
- Guide (patchy wire fabric) wall serves as a separator between Li streams and wire ropes.
- Wire ropes provide the FW force balance.
- Second patchy wire fabric layer separates the wire ropes from FLiBe.
- FLiBe blanket is an element of FW.

Consistency with the FPR is outstanding:

- Excellent energy extraction from the plasma and the blanket.
- Wires can withstand any plasma disruptions.
- Be wire ropes multiply neutrons.
- Minimal amount of high-Z materials.
- Vacuum barrier at the plasma boundary.
- Extremely high reliability, no damage, re-placement on the fly.

Yacht Sail FW provides stationary plasma boundary conditions.

At the same time it is insensitive to thermal deformations.

Yacht Sail FW eliminates the necessity in the stationary tokamak regime.

Development of 1 m^2 of the FW requires 1 kg of tritium

The tritium consumption $W_{T,FW}$ for development of the first wall is straightforward to calculate, and for 15 MW·year/m^2 is given simply by

$$W_{T,FW} = 1.046 \quad \frac{\text{kg}}{\text{m}^2}. \tag{1.13}$$

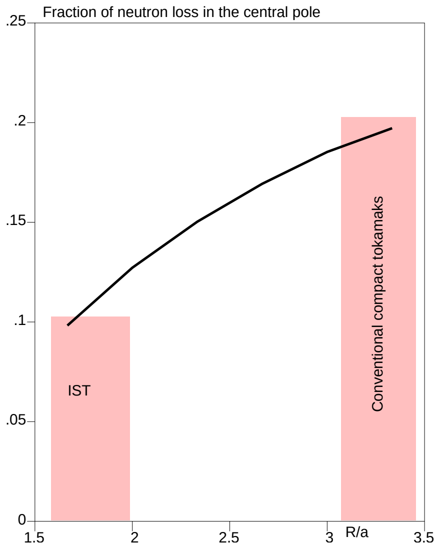
Such a large consumption of tritium automatically requires breeding tritium with efficiency close to or exceeding 100 %.

It makes three elements of magnetic fusion, i.e., OPRR, FW and Tritium Cycle all linked together by the requirement of 100 % tritium breeding starting from the early phase of development of a fusion reactor.

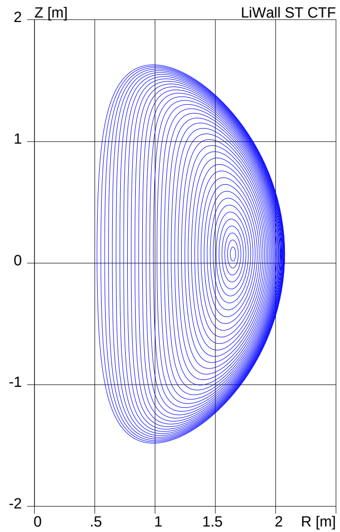
Reactors or reactor size machines are not suitable for such a triple-goal R&D (600-700 kg of tritium for ITER size device)

Compact plasma devices are necessary for developing OPRR+FW+TC

Among compact plasmas, ST are, in fact, the only option



Fraction of neutron losses into the central pole as a function of aspect ratio. (Plasma cross-section of IST was used for calculations.)



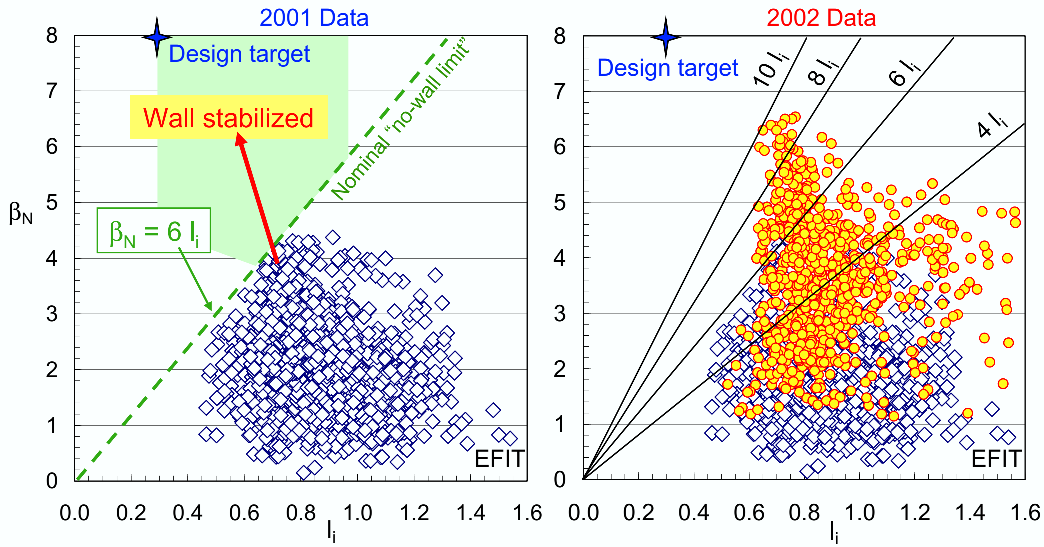
Plasma cross-section used for calculations

There is no reason to keep high-β ST in a sub-critical regime instead of igniting it. From the FW technology point of view,

Ignited ST (rather than driven CTF) suggests use of full FW area for tritium breading

START, NSTX, MAST demonstrated OPRR relevant beta (35 %)

Plasma operation in low l_i, wall-stabilized space



- Normalized beta, $\beta_N = 6.5$, with $\beta_N/l_i = 9.5$; β_N up to 35% over $\beta_{N\ \text{no-wall}}$
- Toroidal beta has reached 35% ($\beta_t = 2\mu_0\langle p\rangle / B_0^2$)

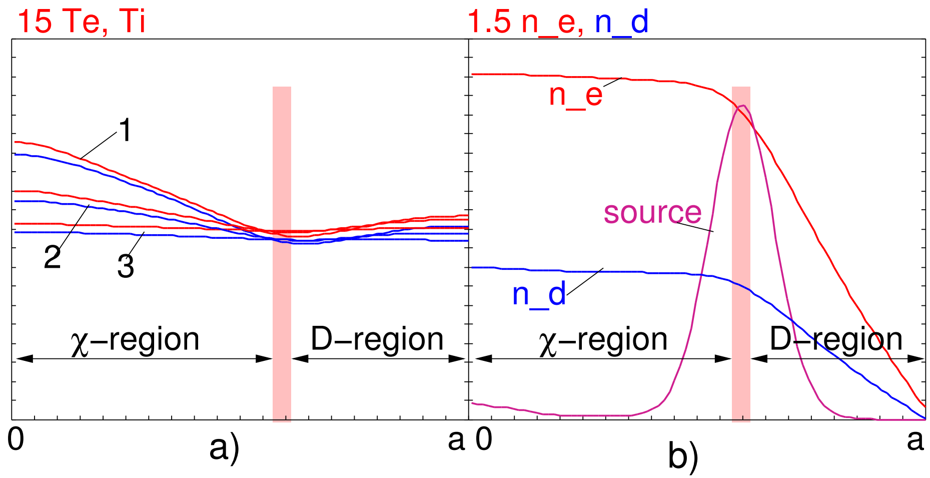
NSTX

In contrast to high-B concept, ST have a relevant stability data base

Absorbing LiWalls result in high edge temperature “pedestal”

$$\Gamma^{micro}_{edge\rightarrow wall} \simeq \Gamma_{core}, \quad T_{edge} = \frac{1}{\gamma\Gamma_{core}} \int P_{heat} dV \simeq T_{core}. \qquad (2.1)$$

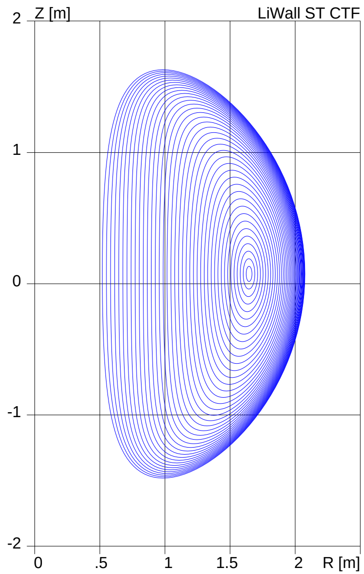
Core fueling lead to separation of thermo-conduction region from the wall and improved confinement (e.g., ITER would ignite in LiWall regime).



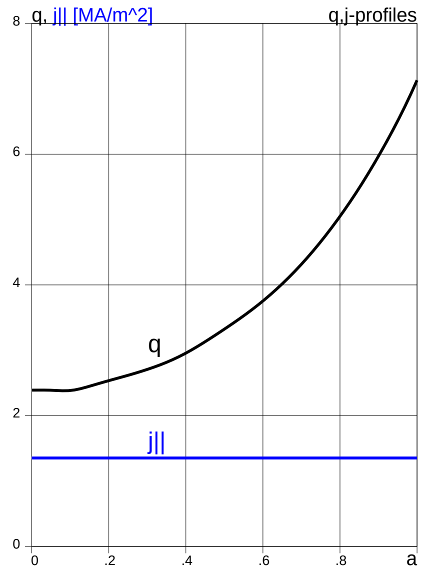
χ- and D- confinement regions in the low recycling regime. (a) Electron and ion temperatures for three values of thermo-conduction coefficients. (b) Electron, ion density and the particle source.

Flattened plasma temperature leads to second stability in the plasma core

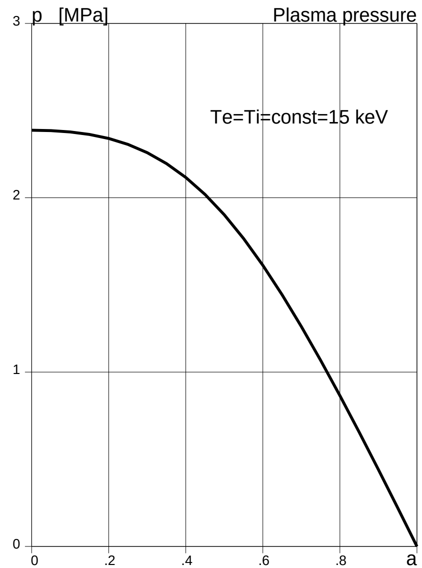
LiWall at the plasma boundary stabilize the free boundary modes



(a)



(b)

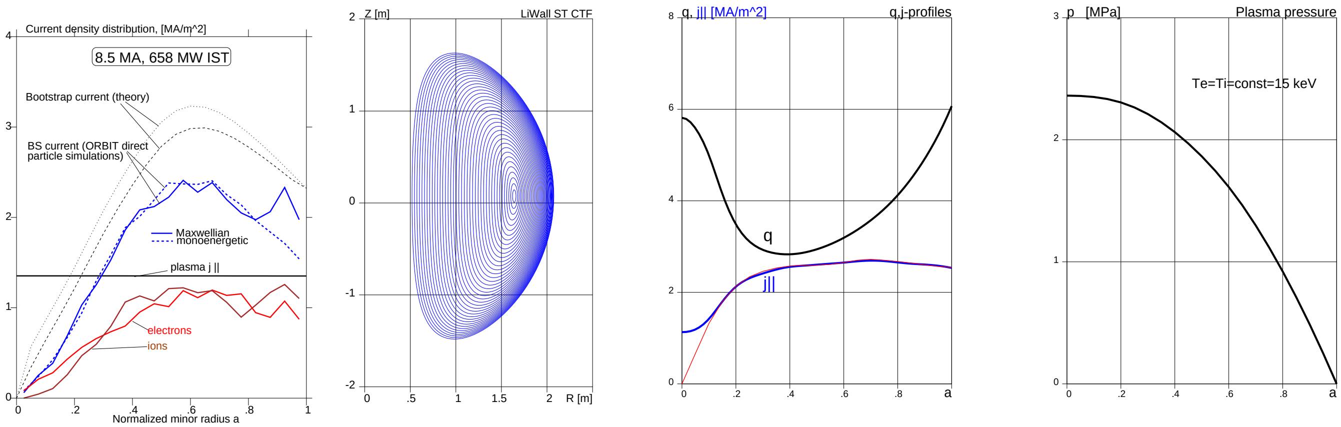


(c)

Figure 3: (a) Stable magnetic configuration of Ignited Spherical Tokamak with $\boldsymbol{I_{pl} = 8.5}$ MA, $\boldsymbol{\beta = 0.46}$. (b) Parallel current density and $\boldsymbol{q}$-profile. (c) Pressure profile (exceeding OPRR level).

Despite low B of ST, the second stability core and high-β allows for OPRR level of $\langle p \rangle$

No profile control is needed for a required bootstrap current in IST



(a) (b) (c) (d)

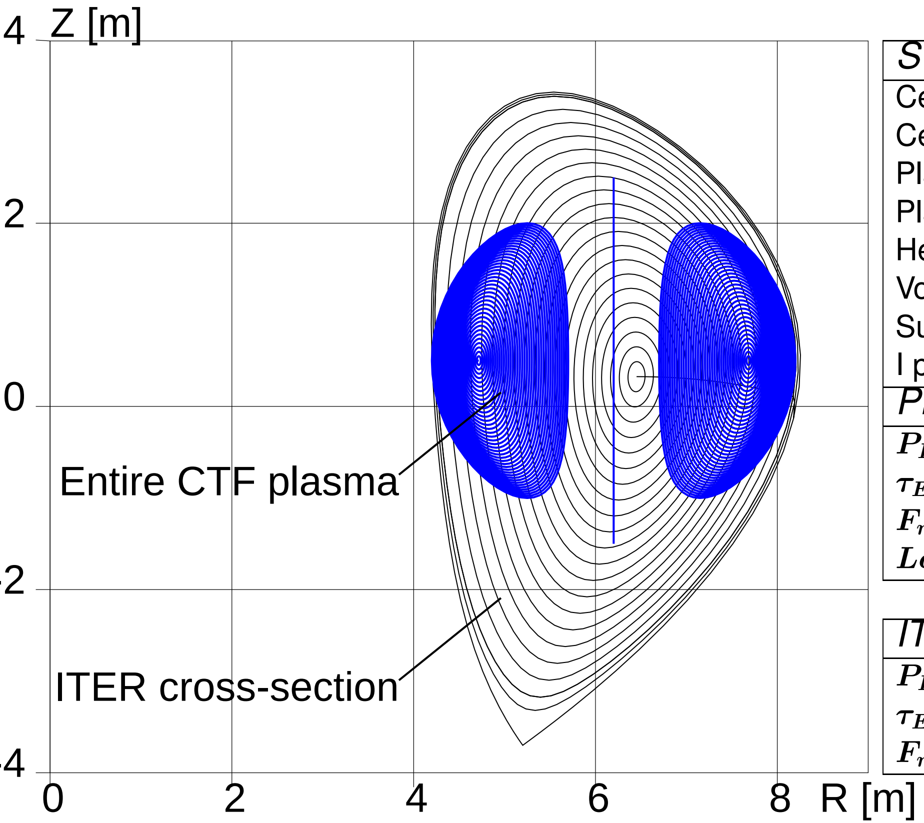
(a) Bootstrap current profile in IST configuration of previous Fig. (b) Stable magnetic configuration of bootstrap current maintained IST configuration with $I_{pl} = 9.2$ MA, $\beta = 0.44$. (c) Parallel current density $j_{\|}$ (blue) aligned with the bootstrap current (red) and q-profile. (d) Pressure profile.

In the LiWall regime (except the very plasma center)

the level of bootstrap current can be expected suffi cient for steady operation of IST

Ignited Spherical Tokamaks (IST) and ITER are two parts of fusion.

Crucial difference is in β=0.4 (vs 0.03) and in "flat" $T_{i,e}$ =15 keV (vs peaked)



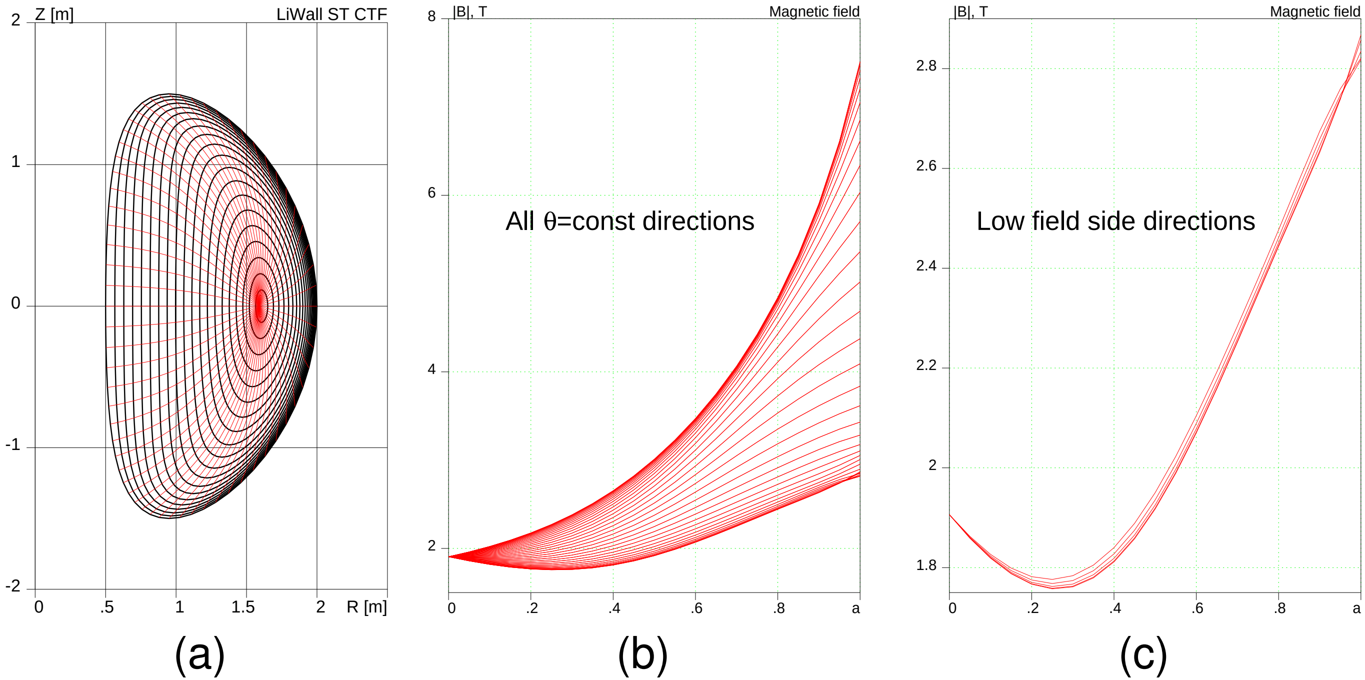
ST CTF Parameters				
CenterPole R	m	0.5	0.5	0.5
CenterPole B	T	7.5	7.5	7.5
Plasma R_1	m	0.5	0.5	0.5
Plasma R_2	m	2.0	2.0	2.0
Height	m	3.0	3.2	3.4
Volume	m^3	26.1	27.8	29.6
Surface	m^2	53.4	55.9	58.5
I plasma	MA	11.1	11.9	12.7
Plasma performance				
P_{DT}	MW	388	490	606
τ_E	sec	0.75	0.69	0.64
$F_{neutron}$	MW/m^2	5.8	7.0	8.3
$Loss_{neutron}$	%	4.7	4.8	4.9

ITER				
P_{DT}	MW	410	V	834 m^3
τ_E	sec	3.7	S	680 m^2
$F_{neutron}$	MW/m^2	0.5		

IST are suitable for developing OPRR and fulfilling the CTF mission related to FW and TC R&D.

Expected unique confinement and stability physics of LiWall ST makes the question on feasibility of DD reasonable

Absolute magnetic well situation can be created in IST

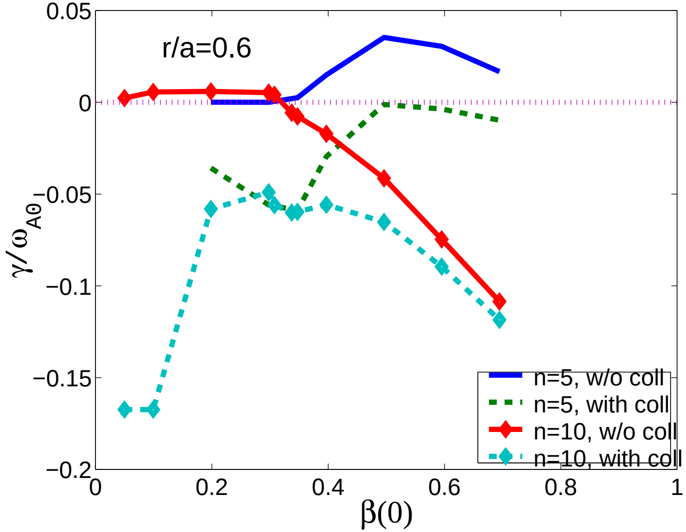


(a) Stable magnetic configuration of Fig. 3 (IST with $I_{pl} = 8.5$ MA, $\beta = 0.46$). Red lines correspond to $\theta = const$. (b) $|B|$ as a function of a for 64 equidistant θ values. (c) $|B|$ as a function of a for 5 θ values near the outer middle plane.

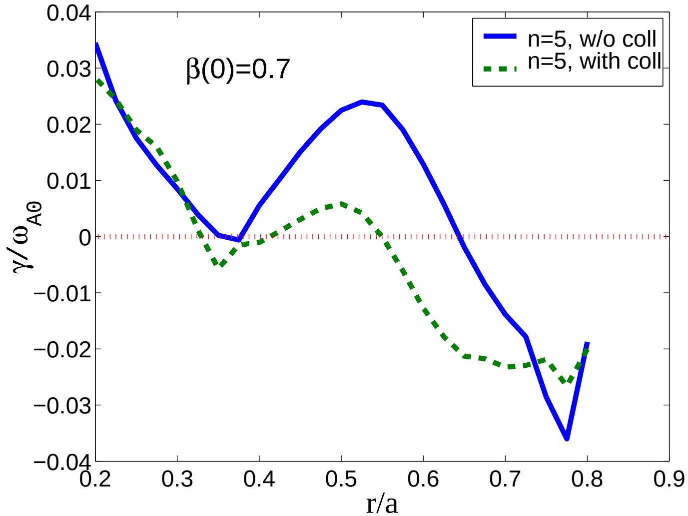
Large inverse gradient of $|B|$ leads to reversing the particle precession

Electron trapped modes are stabilized by reversed particle precession

Gorelenkov's HINST calculations of stability



Increase in β stabilizes modes



Even $n = 5$ mode is stable at $a > 0.6$

In LiWall regime particle losses are determined by the best confined component.

With no micro-turbulence DD fusion might be possible (needs $\tau_E \simeq 20$ sec)

but probably unpractical because of low power density.

The present day situation when 2 sites are ready to proceed with ITER is incredibly good for fusion

For its final goal, i.e., the power reactor, ITER addresses several important topics:

- *It is the first plasma device overpassing break-even and entering the fusion regime*
- *It fulfills many reactor engineering & technology objectives (designing and testing reactor-size tokamak assembly, power and other support, control and data acquisition systems, etc)*
- *ITER will create a unique management and control infrastructure relevant to fusion reactor operation and maintenance,*
- *turn more attention of the society to fusion*
- *. . .*

At the same time,

The green light for ITER, in fact, obligates fusion community to proceed with a reactor R&D

A comparable in scale and complimentary to ITER program is necessary and imminent for fusion

With its low power density ITER cannot contribute substantially to R&D of Operational Power Reactor Regime, First Wall and Tritium Cycle of the reactor.

Making analogy with the space program,

With ITER fusion would only prepare a launching pad for its "space-crafts" (reactors) with no "rocket" (or even its engine) ever been developed.

The "rocket engine" $\equiv$ (high-β OPRR) and the "rocket" $\equiv$ (FW & TC) itself should be prepared by a separate program oriented unavoidably to Ignited ST

Time will come (hopefully in a foreseeable future) of launching it in parallel to ITER (presumably at the second site left after ITER decision).